PRELIMINARY INVESTIGATION OF THE USE OF MONOLITHIC U-MO FUEL IN THE MIT REACTOR

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ABSTRACT

Studies have begun on the use of monolithic LEU U-Mo fuel in the MIT Nuclear Research Reactor (MITR-II) using the Monte Carlo Transport code MCNP. These studies have included model benchmarking, LEU fuel optimization, burnup evaluation, in-core facility design, and determination of safety attributes. Benchmarking studies on the initial core have shown favorable agreement between the calculated and measured reactivity worths of the six control blades. In addition, optimization studies on LEU U7Mo MITR-II fuel have shown that an arrangement of ten to twelve plates per fuel element would have initial reactivity values and thermal neutron fluxes comparable to the current HEU core. Burnup studies which have been made using the MCODE depletion program will be presented. Safety attributes such as temperature coefficients, shutdown margins, and coolant subcooled margin are under evaluation.

1. Introduction

The MIT Research Reactor (MITR-II), currently licensed to operate at 5 MW, contains a hexagonal-shaped core with twenty-seven rhomboid-shaped fuel element positions, as can be seen in Figure 1. Typically, at least three of these positions are filled with either an in-core experimental facility or a solid aluminum dummy element to reduce power peaking. The remaining positions are filled with standard MITR-II fuel elements.

Each of fuel element currently contains fifteen plates of U-Al_x HEU fuel (fuel density of 3.4 g/cm^3), with a fuel thickness of 0.76 mm (0.030 inches) and aluminum cladding with a thickness of 0.38 mm (0.015 inches). In order to increase the heat transfer capabilities of the fuel plates, each plate contains vertical fins of 0.25 mm (0.01 inches) height and 0.25 mm width.

Six boron-stainless steel control blades, located at the periphery of the core, serve as the coarse control of the reactor. Fine control is maintained by an aluminum clad cadmium regulating rod, located at one of the corners of the reactor.

Light water acts as both moderator and coolant of the reactor. An outer tank containing the D_2O reflector surrounds the inner core tank. Beyond the D_2O reflector, a secondary reflector of graphite contains several experimental ports for thermal neutron irradiations.

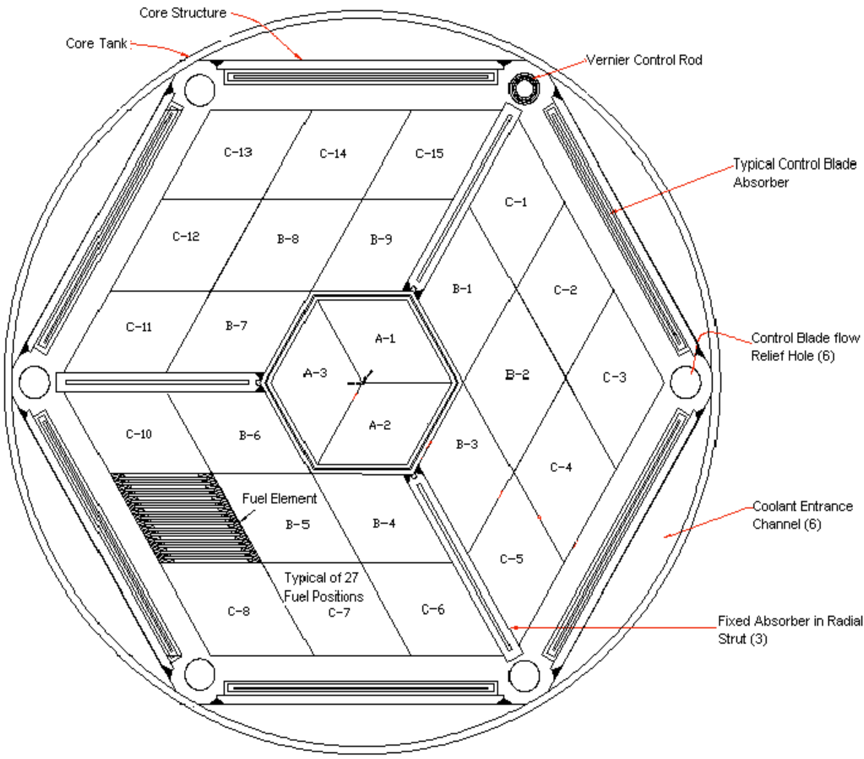

Figure 1. The MITR-II core.

The use of conventional LEU dispersion fuels would not allow the reactor to maintain criticality because of the compactness of the core. Therefore, conversion of the core to LEU had to await development of higher density fuels. The development by the RERTR program [1] of a high-density monolithic U-Mo fuel appears promising, and the LEU studies described here use a monolithic U-Mo fuel.

2. Model Validation

A model of the MITR-II using the Monte-Carlo transport code MCNP was first made by Redmond, *et al.* [2]. Each fuel plate is modeled discretely, as are most of the relevant reactor structures. A variation on this model was used in all calculations for both validation of the HEU core and subsequent LEU studies. Initial validation of the model was made by comparing calculations with measured values of both K_{eff} and fast neutron flux.

The core configuration chosen for validation studies is the second configuration used after the reactor was redesigned in 1975. This configuration used twenty-two fresh fuel elements and five solid aluminum dummies. The reason this core configuration was chosen for validation is because of low burnup in the initial core configuration (less than 2.5 MWd/kg), along with the absence of fixed absorbers in-core, and a relatively long operating period (eight months) over which reactivity data was taken.

In addition to comparing the calculated K_{eff} at a control blade height equivalent to the initial critical (K_{eff}=1) height, blade positions were varied in the model and the resulting reactivity worths compared with the measured blade worth curve.

For fuel burnup calculations, the MCNP model was used with the point depletion code ORIGEN through the linkage code MCODE, developed by Xu, *et al*. [3]. Unlike other coupling codes, MCODE uses a predictor-corrector approach to burnup depletion, as opposed to single timestep evaluations as in MONTEBURNS or MOCUP.

Because of the limitation on the number of points able to be tallied in MCNP (99), each individual plate could not be discretely depleted by the use of MCODE. Instead, each of the twenty-two fuel elements was initially modeled so that all fifteen plates in an element were assumed to have identical material compositions.

The MCODE model was run at a power density of 36 kW/l, which is equivalent to a reactor power level of 2.5 MW, the nominal power under which core #2 was run. Changes in K_{eff} (i.e. reactivity) with burnup were calculated and compared with measured reactivity changes (from blade position changes). In addition, the sensitivity of the burnup calculations to the choice of ORIGEN cross-section libraries was determined.

3. LEU Studies

For the LEU evaluation, monolithic U-Mo fuel with a concentration of 7 w/o Mo (U7Mo) was chosen. This fuel has a density of 17.5 g/cm^3.

The criteria for LEU fuel are that fuel elements would be designed to fit into the current core configuration, and be similar in design to current fuel elements. Design goals for the project are as follows:

- Thermal flux equivalent to or greater than the HEU core at the same power.
- In-core facilities have fast flux at least equivalent to the HEU core.
- Negative moderator temperature and void coefficients.
- Fuel cycle length equivalent to or longer than HEU core.
- Adequate control blade worths and shutdown margins.
- Sufficient excess reactivity to overcome xenon poisoning and Doppler broadening.
- Adequate subcooled margin in all coolant channels, and
- Adequate natural convection cooling for low power and shutdowns.

After validation of the model, U7Mo LEU fuel was substituted for the UAl_x HEU fuel and both the number and fuel thickness of the plates was varied to determine the K_{eff} and neutron fluxes in each configuration. Cladding thickness remained at 0.38 mm (0.015 inches) in all cases. In selected cases, the water density was varied in order to determine an overall moderator reactivity coefficient.

Burnup comparisons were made between HEU core #2 and a similar configuration using eleven and twelve plate elements in an LEU core. Reactivity values were determined at 5 MW over about 300 days of full power operation, about twice the number of days that core #2 was actually operated.

In addition to neutronic calculations, heat transfer coefficients were determined for various fuel designs.

4. Validation Results

The initial critical blade height for core #2 was 21.6 cm (8.5 in.). At this blade height, the MCNP calculated K_{eff} was 0.996 +/- 0.00049, which is in excellent agreement with the measurement. The slightly lower value is possibly attributable to the absence of fuel tapering in the MCNP model.

Reactivity values as compared with blades fully in were taken at blade positions from the critical height to 34.3 cm (13.5 in.) and compared with a measured reactivity worth curve. Because cadmium blades were replaced with boron-stainless steel blades during core #2, both materials were compared. As shown in Figure 2, there is good agreement between the model and actual measurements. In addition, very little reactivity difference can be seen between the two materials.

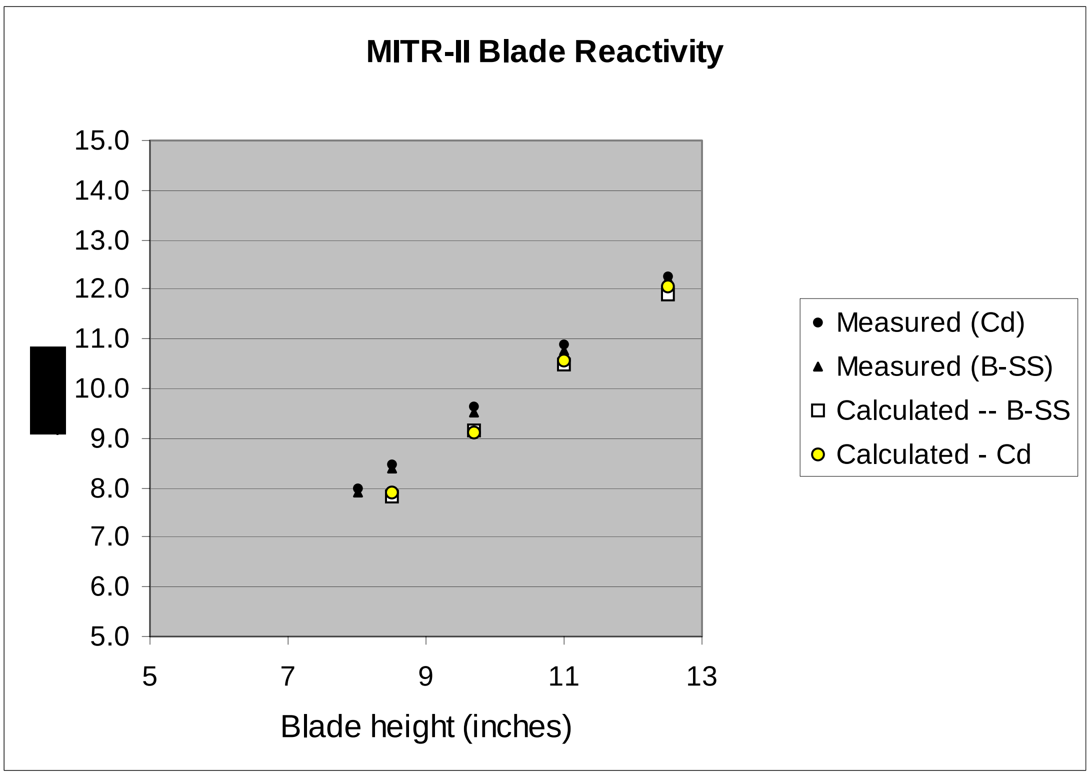

Figure 2. Blade Worths

Burnup calculations from MCODE that were compared with measured reactivity data are shown in Figure 3. These show good agreement with values taken from control blade positions at the end of an operating week, when xenon poisoning is at equilibrium. The slight offset at the zero reactivity point is because of the 2.3 MWd/kg generated in Core #1.

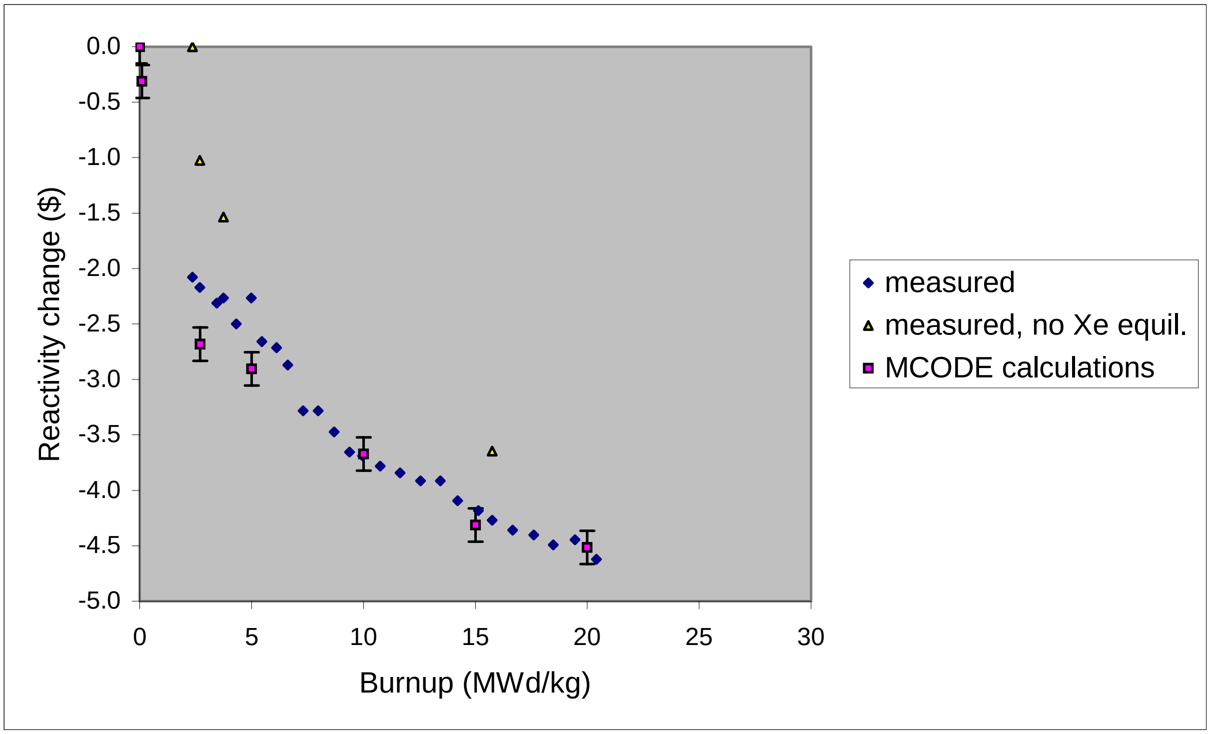

Fig. 3. Reactivity Values of MITR-II Core #2.

The choice of the ORIGEN cross-section library showed no significant difference in reactivity values between the use of a PWR library (PWRU) and that of an LMFBR (FFTFC). Note however that the effective one-group cross sections used for ^{235}U in the ORIGEN depletion are obtained directly from the MCNP calculation, and therefore reflect the local neutron spectra in the MITR-II core.

5. LEU Results

Figure 4 shows a comparison of fuel plate thicknesses for a varied number of plates per element. This shows that in order to obtain criticality at the same blade height as the HEU core, an increase in moderation is needed for an LEU core. This will be obtained through reduction of the number of plates per element from fifteen to thirteen or less. In addition, it will be necessary to reduce the fuel thickness.

The moderator reactivity coefficients were determined by comparing the K_{eff}s with the water density corresponding to that at 20 ^oC to that at 50 ^oC, the normal maximum operating temperature for the MITR-II. Values were determined in most LEU cases, from fifteen plates per element to nine. All values remained negative and showed very little change from the fifteen plate HEU value of -0.015 %ΔK/K ^oC.

Similarly, thermal neutron flux values in the outer fuel ring for various critical configurations are shown in Figure 5. In order to achieve the desired LEU thermal flux, again an increase in moderation is necessary. As seen in both Figure 4 and Figure 5, an LEU fuel element with thirteen plates or less is needed to both increase the thermal flux to the HEU value and maintain criticality.

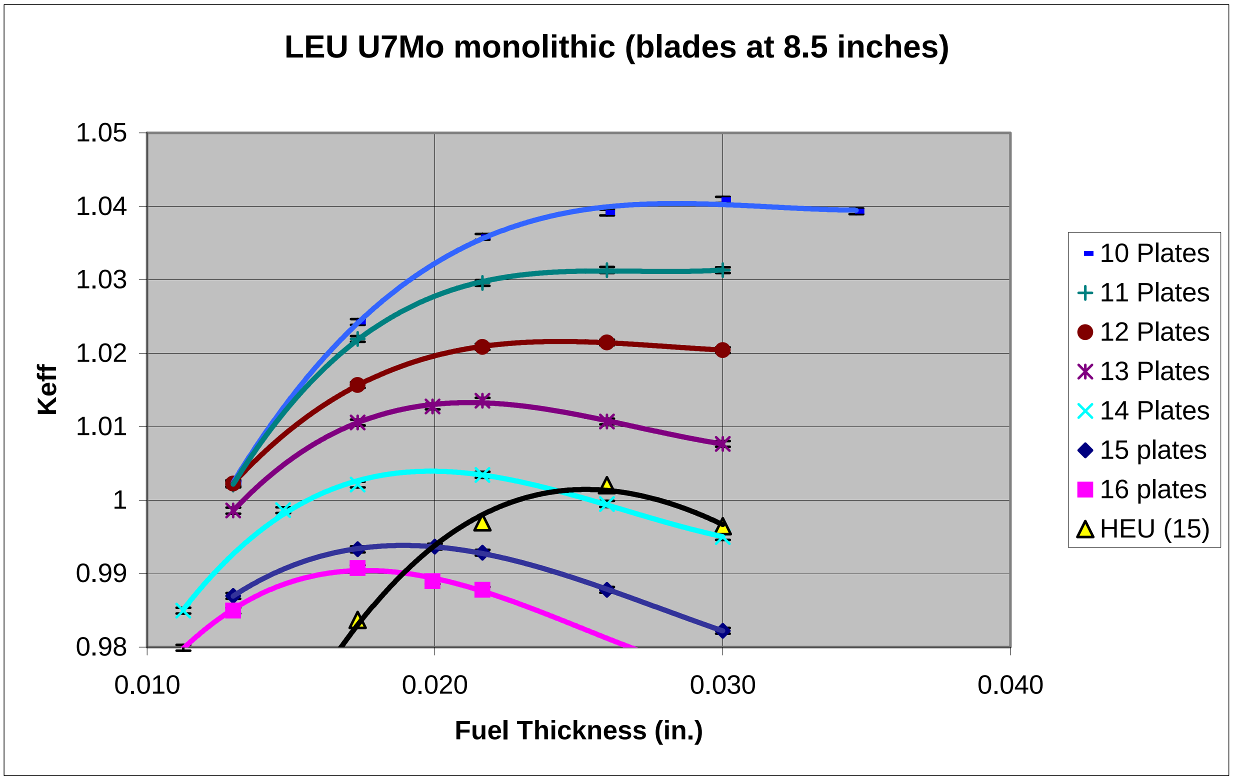

Figure 4. Keff vs. Plate Thickness.

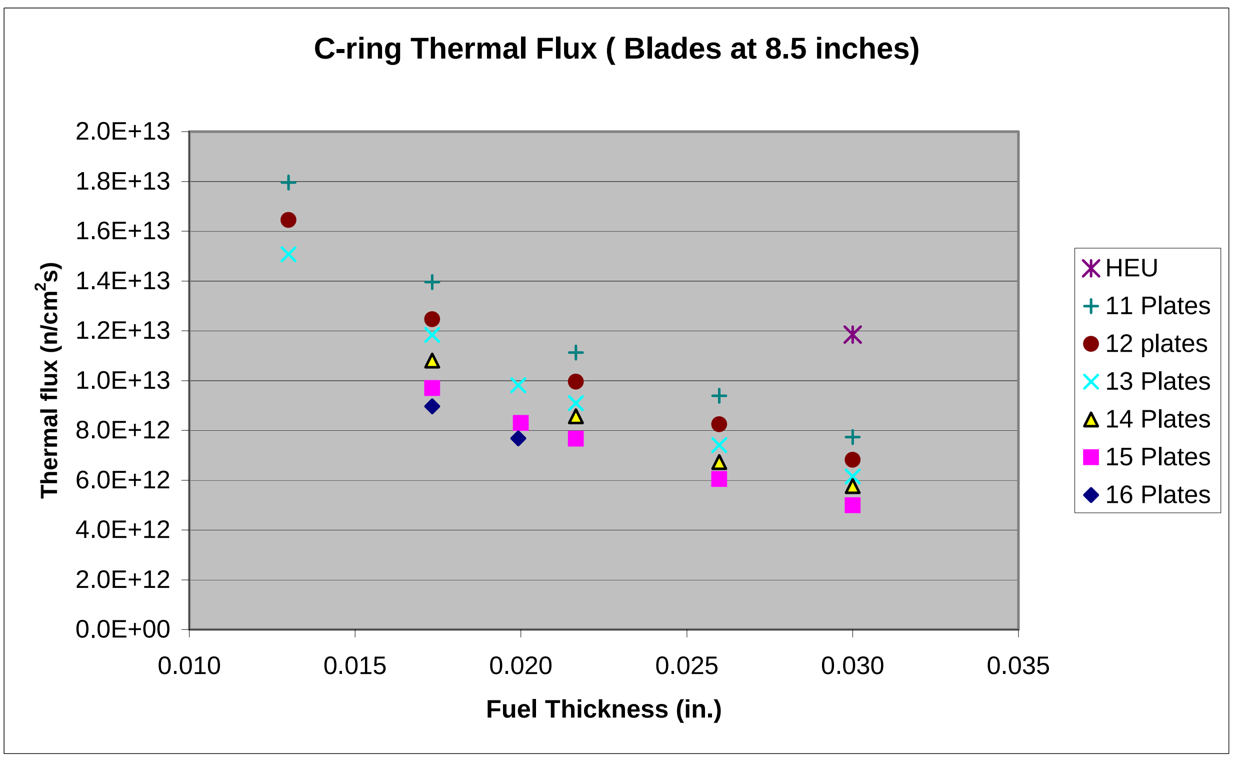

Figure 5. Thermal Fluxes

There has been no LEU configuration identified that would have an in-core fast flux equivalent to that of the HEU core. Table 1 shows a comparison between the current HEU core and a twelve-plate LEU core. An in-core sample facility will be designed to enhance the fast flux for experimental needs. One possible design, which is to use a 1.3 mm annulus of LEU fuel, is presented in Table 1. This shows a fast flux enhancement of nearly a factor of two. Other designs to approximate various reactor spectra are being considered.

Further study is needed to determine the flux changes in other facilities, such as the Fission Converter Facility, used in Neutron Capture Therapy research.

	In-Core Sample Area Flux (n/cm^2s)		
	Thermal (<0.4 eV)	Epithermal (0.4 eV-3KeV)	Fast (>3KeV)
HEU – current design	5.92E+13	4.75E+13	8.65E+13
LEU – current design	5.62E+13	4.27E+13	7.80E+13
LEU 0.13 cm thick fueled annulus	2.34E+13	6.23E+13	1.71E+14

Table 1. In-core sample facility fluxes

Burnup reactivity changes as compared with the initial (fresh) cores are shown in Figure 5. Given the similarities of the neutron fluxes in the eleven plate LEU core to the HEU core, burnup values are very similar. A slightly smaller reactivity drop of the LEU at higher burnups is because of the increase in ^{239}Pu from the larger amount of ^{238}U in the LEU.

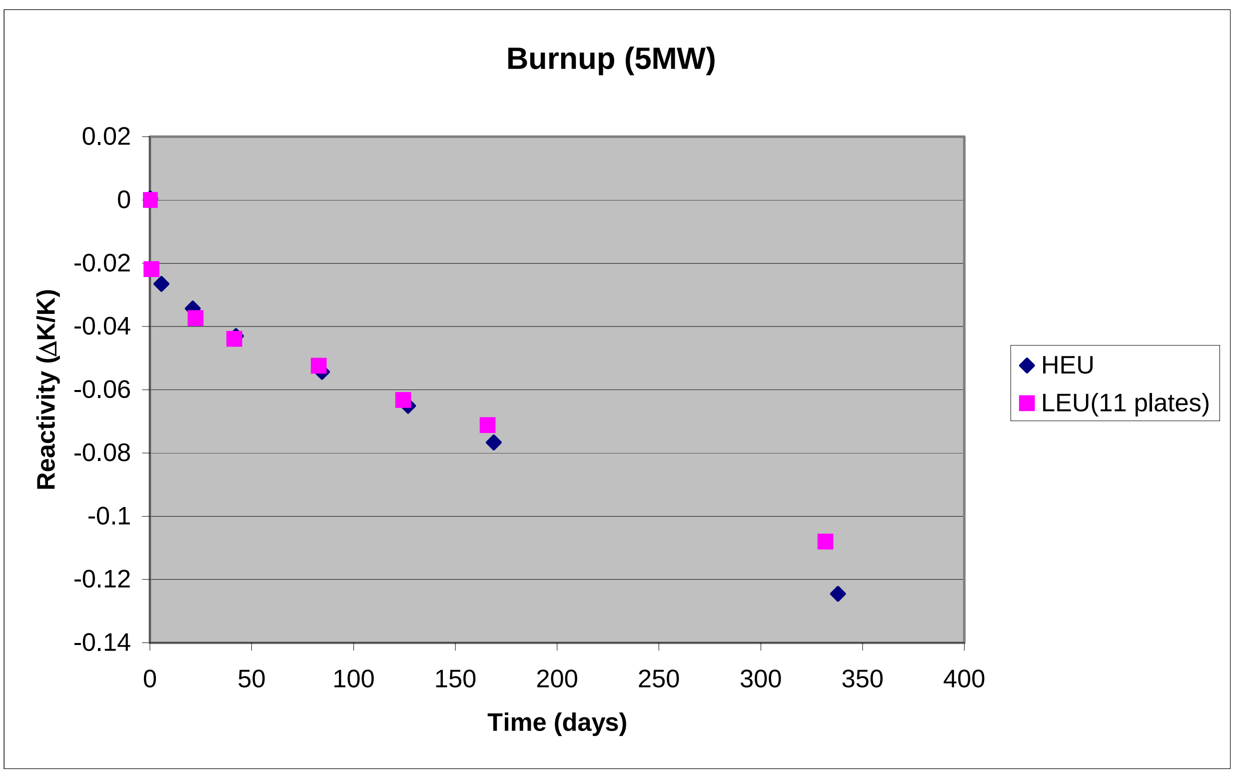

Figure 5. Burnup

Preliminary heat transfer and thermal-hydraulic parameters calculated for configurations corresponding to their peak K_{eff} are shown in Table 2. These are taken at a nominal operating power of 5 MW and a typical primary flow rate of 7570 l/min (2000 gal/min). Because of an increase in the amount of heat generated per plate and despite larger flow areas, there are increased temperatures in the elements with a lesser number of plates. If an element of twelve plates or less is chosen, the coolant flow rate must be increased about 50% in order to avoid nucleate boiling. This would appear feasible using the current primary flow system. However, pressure limits on the reactor core tank would limit the maximum flow from being increased further. This would limit possible future increases in the reactor licensed power level. Calculations involving shutdown cooling using natural convection will be undertaken.

5 MW, 7570 l/min							
No. of plates/element	15 -HEU	15-LEU	14	13	12	11	10
Average Q (W/plate)	13890	13890	14880	16030	17360	18940	20830
1/2 fuel thickness (mm)	0.381	0.254	0.279	0.330	0.330	0.330	0.381
ΔT across plate (^{o}C)	2.35	3.46	4.00	4.95	5.36	5.85	7.27
Water channel thickness (cm)	0.249	0.274	0.298	0.321	0.359	0.405	0.450
Avg. channel mass flow rate (g/s)	346	346	371	399	432	472	519
Reynold's number	20100	20000	21400	23000	24700	26800	29300
Heat transfer coefficient (W/m^{2} ^{o}C)	14300	13000	12700	12400	11800	11300	11000
Film Temperature (^{o}C)	76.0	78.6	81.5	84.4	89.1	94.8	100.6

Table 2. Thermal-Hydraulic Parameters

6. Conclusions

Preliminary calculations show that the MITR-II can be successfully operated using monolithic U7Mo LEU fuel. However, in order to maintain criticality and produce a useful thermal flux, increased moderation will be necessary. This can be achieved by decreasing the number of plates per fuel element from the current fifteen to about twelve and by reducing the fuel thickness. Moderator reactivity coefficients have been calculated to remain negative under normal operating conditions. A decrease in the number of plates would necessitate an increase in the required primary coolant flow.

Burnup calculations using MCODE show similar reactivity changes between the HEU core and an eleven element LEU core.

Design of in-core facilities to enhance the available fast flux is in progress. This will include experimental design to simulate various reactor spectra to be used for materials irradiations. Similarly, further study of flux changes in other facilities is also in progress.

Although further design studies are needed to finalize a fuel design, a successful demonstration of the viability of monolithic fuel by the RERTR program will allow the MITR-II to use LEU fuel as part of its future use as an essential tool in nuclear research.

References

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2. E. REDMOND, J. YANCH, and O. HARLING, "Monte Carlo Simulation of the MIT Research Reactor," *Nuclear Technology*, **106**, pp. 1-14 (1994).
3. Z. XU, P. HEJZLAR, M. DRISCOLL, and M. KAZIMI, "An Improved MCNP-ORIGEN Depletion Program (MCODE) and its Verification for High-Burnup Applications," PHYSOR, Seoul, Korea, Oct. 2002.